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TECHNICAL REPORT

Achievement of Reactor-Outlet Coolant Temperature of 950°C in HTTR

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A High Temperature Gas-cooled Reactor (HTGR) is particularly attractive due to its capability of producing high-temperature helium gas and to its inherent safety characteristics. The High Temperature Engineering Test Reactor (HTTR), which is the first HTGR in Japan, achieved its rated thermal power of 30 MW and reactor-outlet coolant temperature of 950°C on 19 April 2004. During the high-temperature test operation which is the final phase of the rise-to-power tests, reactor characteristics and reactor performance were confirmed, and reactor operations were monitored to demonstrate the safety and stability of operation. The reactor-outlet coolant temperature of 950°C makes it possible to extend high-temperature gas-cooled reactor use beyond the field of electric power. Also, highly effective power generation with a high-temperature gas turbine becomes possible, as does hydrogen production from water. The achievement of 950°C will be a major contribution to the actualization of producing hydrogen from water using the high-temperature gas-cooled reactors. This report describes the results of the high-temperature test operation of the HTTR.

KEYWORDS: HTTR, HTGR, VHTR, 950°C, high temperature, rise-to-power, performance test, inherent safety, helium, hydrogen production

I. Introduction

The High Temperature Engineering Test Reactor (HTTR), which is the first High Temperature Gas-cooled Reactor (HTGR) in Japan has a thermal power of 30 MW and 950°C maximum reactor-outlet coolant temperature, was constructed at the Oarai Research Establishment of Japan Atomic Energy Research Institute (JAERI) for the purpose of establishing and improving technologies of HTGRs as well as nuclear heat utilization.[1)] The HTTR attained its first criticality in November 1998. The rise-to-power tests were started in September 1999 and during the fourth phase of the tests the HTTR reached its full power of 30 MW with reactor-outlet coolant temperature of 850°C in December 2001.[2,3)] In March 2002, JAERI received a pre-operation test certificate, which is an operation permit for the HTTR in the rated operation mode (operation at a reactor-outlet coolant temperature of 850°C) from the government. After receiving the operation permit, the safety demonstration tests[4,5)] were conducted to demonstrate inherent safety features of the HTGRs as well as to obtain the core and plant transient data for validation of safety analysis codes and for establishment of safety design and evaluation technologies not only for the commercial HTGRs but also the VHTR which is one of the Generation IV reactor candidates. The operation history of the HTTR is shown in **Table 1**.

A high-temperature test operation as the fifth and final phase of the rise-to-power tests was conducted to achieve the rated thermal power of 30 MW and reactor-outlet coolant temperature of 950°C after safety evaluations of the fuel, the control-rods and the intermediate heat exchanger for the high-temperature test operation.[6)] Since HTTR is the first HTGR in Japan, it was necessary that the plan was based on the operation data obtained from previous rise-to-power tests for its safety and stability of operations.

The operation in single-loaded operation mode, which used the pressurized water cooler for the primary heat exchanger, was started on 31 March 2004. The reactor power was increased step by step and the reactor-outlet coolant temperature of 950°C was confirmed on 19 April 2004. During the high-temperature test operation, reactor characteristics and reactor performance were confirmed, and reactor operations were monitored to demonstrate the safety and stability of operation.

II. Outline of the HTTR

As the HTTR is the first HTGR in Japan and a test reactor, it has following purposes:

(1) Establishment of basic HTGR technologies
(2) Demonstration of HTGR safety operations and inherent safety characteristics
(3) Demonstration of nuclear process heat utilization
(4) Irradiation of HTGR fuels and materials in an HTGR core condition
(5) Provision of testing equipment for basic advanced studies.

The reactor core, composed of graphite blocks, is so designed as to keep all specific safety features. In the cooling system, the intermediate heat exchanger (IHX) is equipped to supply high-temperature helium gas to some process heat application system being coupled to the HTTR in the future.

The detailed HTTR design was already reported[1)] and the equipments concerning the rise-to-power test are described in this chapter.

1. Core Components and Reactor Internals

The HTTR has a thermal power of 30 MW and 950°C maximum reactor-outlet coolant temperature. The main

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Table 1 HTTR operation history

Operation name	Operation mode		Max. reactor thermal power (MW)	Duration	Operation time (MWd)	Remarks
1st phase rise-to-power test (PT-1)	●	▲	9	23 April–8 May 2000		
				11–26 May 2000	193	
		◆	9	30 May–6 June 2000		
2nd phase rise-to-power test (PT-2)	●	▲	20	3–8 July 2000	63	a)
				29 January–12 February 2001	423	
		◆	20	16 February–1 March 2001		
3rd phase rise-to-power test (PT-3)	■	▲	20	14 April–7 May 2001		
		◆	20	11–16 May 2001	689	b)
				21 May–8 June 2001		
4th phase rise-to-power test (PT-4)	●	▲	30	23 October–14 December 2001	1,293	Achievement of 850°C, 30 MW
		◆	30	25 January–6 March 2002	984	Operation permit issued.
1st operation cycle (RP/RS-1)	●	◆	30	30 May–17 June 2002	370	
		▲	9	21 June–1 July 2002	92	Safety demonstration test
2nd operation cycle (RS-2)	●	▲	30	5 February–14 March 2003	658	Safety demonstration test
3rd operation cycle (RP-3)	●	◆	20	16–21 May 2003	72	a)
4th operation cycle (RS-4)	●	▲	9	8–11 August 2003	25	Safety demonstration test
5th operation cycle (RS-5)	●	▲	30	27 anuary–25 February 2004	549	Safety demonstration test
			18	29 February–5 March 2004		Safety demonstration test
5th phase rise-to-power test (PT-5)	■	▲	30	31 March–1 May 2004	679	Achievement of 950°C, 30 MW
		◆	30	2 June–2 July 2004	626	Operation permit issued.
Total operation time (include other operations, *i.e.* core physics tests)					6,717	

●: Rated operation mode, ■: High-temperature test operation mode, ▲: Single-loaded operation mode, ◆: Parallel-loaded operation mode
a) Automatically reactor shut-down caused by a signal of PPWC flow-rate low
b) Automatically reactor shut-down caused by a loss of off-site electric power by a thunderbolt

Table 2 Major specifications of the HTTR

Item	Specification
Thermal power	30 MW
Coolant	Helium gas
Reactor-outlet coolant temperature	850°C (Rated operation mode)
	950°C (High-temperature test operation mode)
Reactor-inlet coolant temperature	395°C
Primary coolant pressure	4.0 MPa
Primary coolant flow-rate	12.4 kg/s (Rated operation mode)
	10.2 kg/s (High-temperature test operation mode)
Core structures	Graphite
Core height	2.9 m
Core diameter	2.3 m
Power density	2.5 MW/m^3
Fuel	Low enriched UO_2
Enrichment	3–10 wt% (avg. 6 wt%)
Fuel element type	Prismatic block
Pressure vessel	Steel (2·1/4Cr–1Mo)
Number of main cooling loop	1

specifications of the HTTR are shown in **Table 2**.

The reactor consists of a reactor pressure vessel, fuel elements, replaceable and permanent reflector blocks, core restraint mechanism, control-rods, *etc*. Thirty columns of fuel blocks and seven columns of control-rod guide blocks form the reactor core, called the fuel region, which is surrounded by replaceable reflector blocks and large-scale permanent reflector blocks. The fuel element of the HTTR is a pin-in-block type. Enrichment of ^{235}U is 3 to 10 (average 6) wt%. Sixteen pairs of control-rods in the fuel and replaceable reflector regions of the core control reactivity of the HTTR. A control-rod drive mechanism drives each pair of control-rods using an AC motor. At a reactor scram, electromagnetic clutches of the control-rod drive mechanisms are separated, and the control-rods fall into holes in the control-rod guide blocks by the force of gravity at a constant

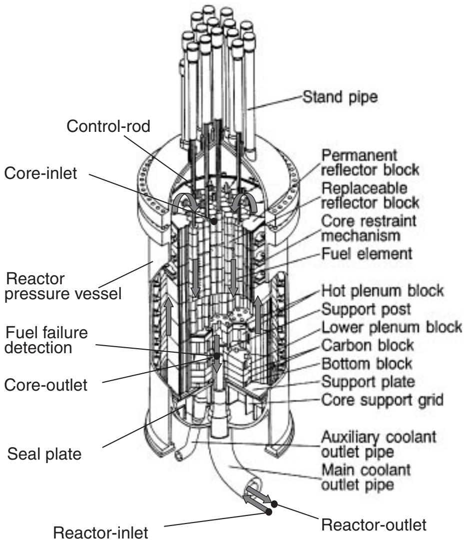

Fig. 1 Vertical cross section of the HTTR reactor

speed, shutting down the reactor safely. The vertical cross section of the HTTR reactor is shown in **Fig. 1**.

2. Main Cooling System

As shown in **Fig. 2**, the cooling system of the HTTR consists of a main cooling system operating at normal operation; and an auxiliary cooling system and a vessel cooling system, the engineered safety features, operating after a reactor scram to remove residual heat from the core. The main cooling system, which consists of a primary cooling system, a secondary helium cooling system, and a pressurized water cooling system, removes heat generated in the core and dissipates it to the atmosphere by a pressurized water air cooler. The primary cooling system consists of an IHX, a primary pressurized water cooler (PPWC), a primary concentric hot gas duct, *etc.* Primary coolant of helium gas from the reactor at 950°C maximum flows inside the inner pipe of the primary concentric hot gas duct to the IHX and PPWC. The primary helium is cooled to about 400°C by the IHX and PPWC and returns to the reactor flowing through the annulus between the inner and outer pipes of the primary concentric hot gas duct. The HTTR has two operation modes. One is the single-loaded operation mode using only the PPWC for the primary heat exchange. Almost all the basic performance of the HTTR system is confirmed by the single-loaded operation mode. The safety demonstration test[4,5] was performed for the purpose of demonstrating HTGR safety operations and its inherent safety characteristics mainly in the single-loaded operation mode. The other is the parallel-loaded operation mode using the PPWC and IHX. In a single-loaded operation mode the PPWC removes 30 MW of heat and in a parallel-loaded operation mode the PPWC and IHX remove 20 MW and 10 MW, respectively. It is planned to use the secondary helium gas of the IHX for nuclear process heat utilization. The auxiliary cooling system, consisting of an auxiliary helium cooling system, an auxiliary water cooling system, a concentric hot gas duct, *etc.* is in stand-by during normal operation and starts up to remove residual heat after a reactor scram. The vessel cooling system cools the biological concrete shield surrounding the reactor pressure vessel at normal operation, and removes heat from the core by natural convection and radiation outside the reactor pressure vessel under ‘accident without forced cooling’ conditions such as a rupture of the primary concentric hot gas duct, when neither the main cooling system nor the auxiliary cooling system can cool the core effectively.

3. Instrumentation and Control System

The reactor power control device consists of control systems for the reactor power and reactor-outlet coolant temperature. These control systems are cascade-connected: the latter control system ranks higher to give demand to the reactor power control system. The signals from each channel of the

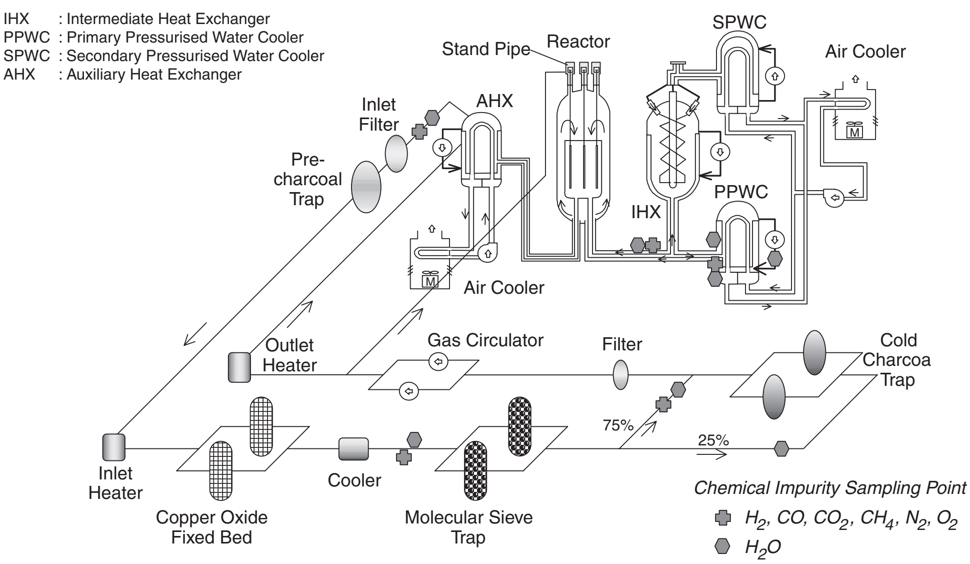

Fig. 2 Schematic diagram of the reactor cooling systems consisting the main cooling system, auxiliary cooling system and vessel cooling system of the HTTR

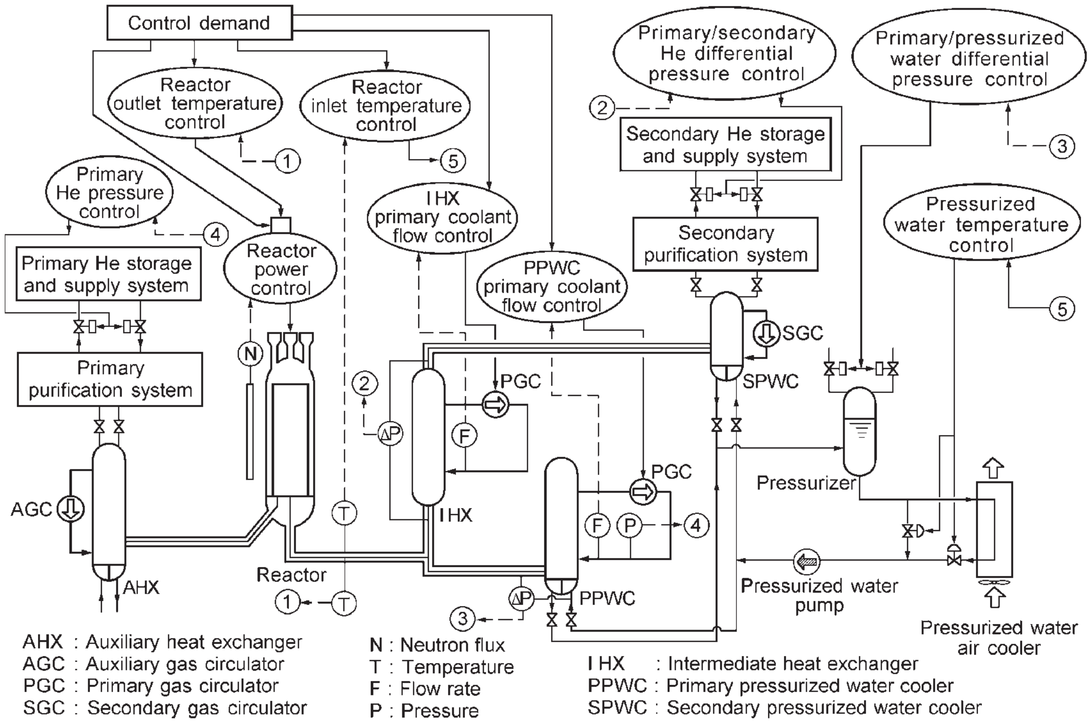

Fig. 3 Plant control device of the HTTR

power-range monitoring system are transferred to three controllers using microprocessors. In the event of a deviation between the process-value and set-value, a pair of control-rods is inserted or withdrawn at the speed from 1 to 10 mm/s according to the deviation. The relative position of 13 pairs of control-rods, except for 3 pairs of control-rods used only for a scram, are controlled within 20 mm of one another by the control-rod pattern interlock to prevent any abnormal power distribution. The plant control device controls plant parameters such as the coolant temperature of the reactor-inlet, flow-rate of the primary coolant, pressure of the primary coolant, and differential pressure between the primary cooling system and pressurized water cooling system. The schematic diagram of the plant control device is shown in **Fig. 3**. The reactor power, the reactor-inlet coolant temperature, and the primary coolant flow-rate are controlled to constant values by each control system. The reactor-outlet coolant temperature is adjustable by the control system of the primary coolant flow-rate.

III. High-Temperature Test Operation

1. Summary

The high-temperature test operation was conducted as the fifth and the final phase of the rise-to-power tests in order to achieve the rated thermal power of 30 MW and reactor-outlet coolant temperature of 950°C. Prior to operation some of the concerns for the high-temperature test operation *i.e.* the maximum fuel temperature, the temperature of control-rods, were confirmed and calculated based on the actual data observed previously by the rated operations.[6] During the operation each parameter was compared with the value measured in the previous high-temperature test operation at the reactor power of 20 MW performed at the third phase of rise-to-power test. The reactor power was increased step-by-step with monitoring all of the parameters, *i.e.* thermal parameters, coolant impurities. The temperature was raised within the rate of 15°C/h (reactor-outlet coolant temperature above 650°C) and 35°C/h (below 650°C) for the safety of operations. The reactor power was kept at 50% (15 MW), 67% (20 MW), and 100% (30 MW) more than two days in a steady temperature condition in order to measure the power coefficients of the reactivity. The reactor power was also kept at 82%, at which the reactor-outlet coolant temperature is a little below 800°C, in order to remove the chemical impurity by a helium purification system. The calibration of the neutron instrumentation system with the thermal reactor power was performed at 97% power.

The high-temperature test operation started on 31 March 2004. **Figure 4** shows the operation history of the reactor-inlet and -outlet coolant temperature and the reactor power. The reactor-outlet coolant temperature of 950°C was achieved on 19 April 2004 during the single-loaded operation mode. During the parallel-loaded operation mode the reactor-outlet coolant temperature reached 941°C and the secondary helium temperature at the IHX-outlet reached 859°C on 24 June 2004. The differences of the reactor-outlet coolant temperature from the design value of 950°C were caused by a permitted margin for error of the flow-rate indicators of the primary cooling system. The attained temperatures implied that the flow-rate of the parallel-loaded operation mode was about 1% higher than that of the single-loaded. As the flow-rate was designed to keep its control target constantly, some correction will be added to the target value of the primary coolant flow-rate in next parallel-loaded operations.

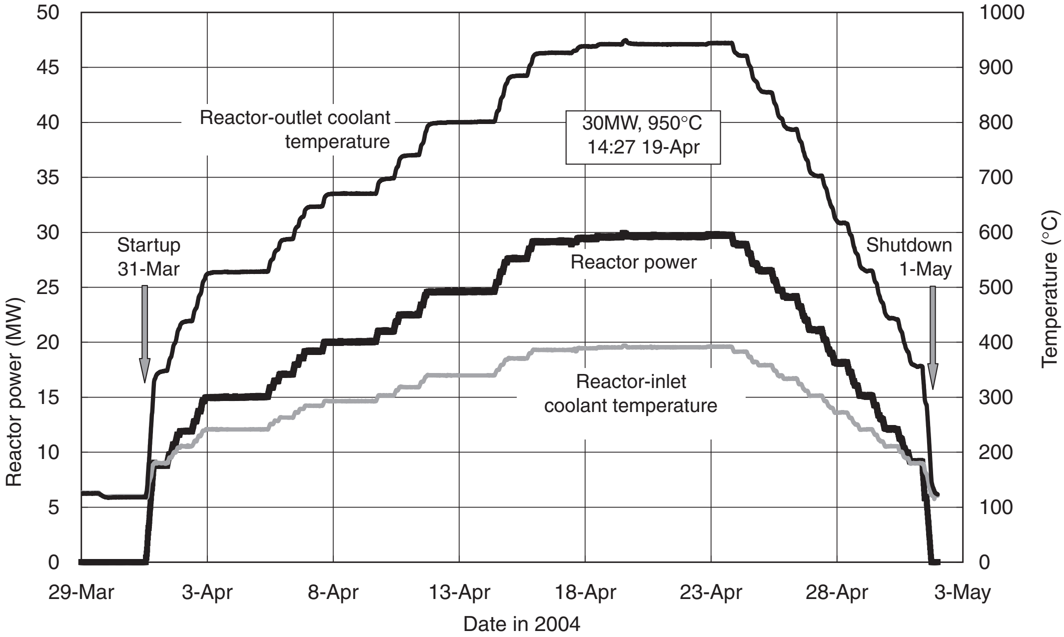

Fig. 4 Operation history during the high-temperature test operation by the single-loaded mode
Maximum reactor-outlet coolant temperature of 950°C had achieved at 14h27 on 19 April 2004

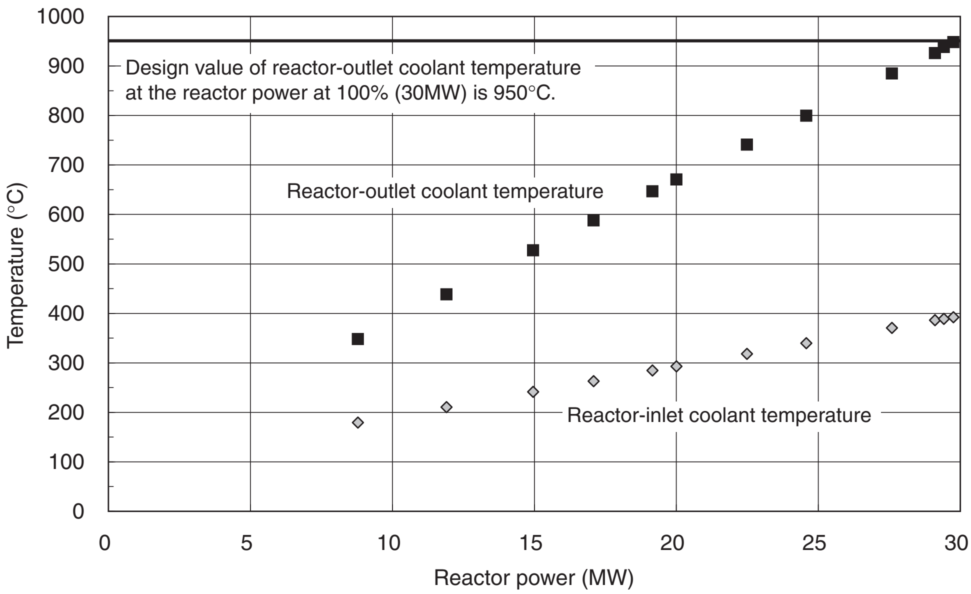

Fig. 5 Relation between the reactor power and reactor-inlet and -outlet coolant temperature

The total operation time of the HTTR was 6,717 MWd (224 EFPD). The preparation for the connection of some nuclear process heat utilization system to the HTTR will be started in the near future.

After confirming the maximum values of temperature and pressure of the primary coolant, *etc.* by the government, JAERI received a pre-operation test certificate, which is an operation permit for the HTTR in the high-temperature test operation mode from the government.

The following are the major test results obtained during the high-temperature test operation.

2. Reactor-outlet Coolant Temperature and Heat Balance

The maximum values of temperature and pressure of the primary coolant which were measured during steady state conditions at the full power of 30 MW operation were confirmed to be less than the criteria of 957°C and 4.0 MPa. **Figure 5** shows the relation between the reactor thermal power and the reactor-inlet and -outlet coolant temperature.

The heat transfer performance was estimated in order to confirm the performance of the PPWC and air fin cooler (ACL). The PPWC is the only primary heat exchanger used at the primary circuit during the single-loaded operations and

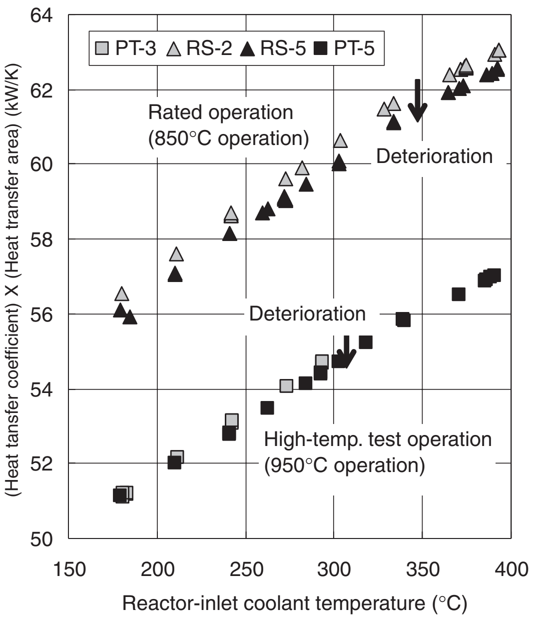

Fig. 6 Heat transfer performance of the PPWC

the ACL is the final heat sink during any operations excluding after a reactor shutdown. **Figure 6** shows the heat transfer performance of the PPWC plotted by 'PT-5' with the estimated values calculated using the data of the third phase of rise-to-power test 'PT-3' conducted in the high-temperature test operation mode in April 2001 and the rated-power and single-loaded operations 'RS-2' in February 2003 and 'RS-5' in February 2004. The heat transfer performance of PPWC during PT-5 operation becomes a little lower than that of PT-3 operation; also the heat transfer was decreased from RS-2 operation to RS-5 operation. It can be considered that the decline of the heat transfer performance depending on the operation time of the PPWC is caused by some mill scale formation on the heat transfer tube or some scale adhesion. The pressurized-water temperature at the PPWC-outlet was 138°C and the reactor-inlet and -outlet coolant temperatures were 392°C and 941°C, respectively. The actual heat transfer performance of about 12% lower than that of the design value was within the assumed limits. The ACL-inlet and -outlet air temperatures were 19°C and 79°C, and the ACL-inlet and -outlet pressurized-water temperatures were 138°C and 69°C, respectively. The actual heat transfer performance of about 9% lower than that of the design value was within the assumed limits.

3. Coolant Chemistry

Chemistry control is important for the helium coolant because impurities cause oxidation of the graphite used in the core and corrosion of high-temperature materials used in the heat exchanger, *etc.* The coolant chemistry was monitored by the helium sampling system continuously between the reactor start-up and shut-down. The actual chemical impurities of carbon monoxide, hydrogen, carbon dioxide, water vapour, methane, nitrogen, oxygen, *etc.* were removed by the helium purification system. The concentration of each impurity was extremely limited by the operating procedure during the operation. The impurity limits of the primary coolant of the HTTR are shown in **Table 3**.

Figure 7 shows the chemical impurity behaviour at the reactor-inlet during the single-loaded and high-temperature test operation period. Each impurity was steadily removed by the purification system. In the operations below 850°C which were previously performed, impurities did not increase rapidly. However, after the temperature rose from 850°C, the impurities of hydrogen, carbon monoxide, carbon dioxide and nitrogen increased rapidly, and small amounts of methane and oxygen were detected. There are the two reasons for the increase: One is the impurity emission from the graphite material used in the core and as an insulator in the concentric hot gas duct. The other is the chemical equilibrium in the core. The water vapour which was emitted from the graphite converted to hydrogen and carbon monoxide by an immediate reaction in the high-temperature conditions in the core. Therefore, the behaviour of hydrogen and carbon monoxide were very similar to that of water vapour especially after the power up from 850°C.

4. Thermal Hydraulics

The core-internal thermal-hydraulic performance of fuel temperature, core-internal structure, and core-internal coolant distribution were confirmed to be appropriate to their design during the full power operation. **Table 4** shows the core-internal structure temperature. The maximum temperature of the core support-plate measured at the upper surface of the centre core support-plate was 450°C which was sufficiently below its limited value of <530°C. Also, it was confirmed that other core-internal structure temperatures were well below their design criteria. From the result that no core-internal structure temperature measurement showed an abnormal value, it was confirmed that there was no abnormal leak flow of coolant such as cross and bypass flows between fuel blocks, replaceable reflector block, permanent reflector block, *etc.* The maximum fuel temperature was evaluated to be 1,463°C prior to the high-temperature operation.[6] It was re-evaluated using the measured temperature data *i.e.* core-inlet and -outlet coolant temperatures shown in **Table 5** and the calculated value of 1,478°C does not exceed the normal operation limit of 1,495°C.

5. Power Coefficients of Reactivity

A negative power coefficient of the reactivity is required for self-stabilization of reactor operations and defined as

Table 3 Chemical impurity limits of primary coolant at temperature above 800°C

Chemical impurity at reactor-inlet	H_2	CO	H_2O	CO_2	CH_4	N_2	O_2
Concentration (volume ppm)	3.0	3.0	0.2	0.6	0.5	0.2	0.04

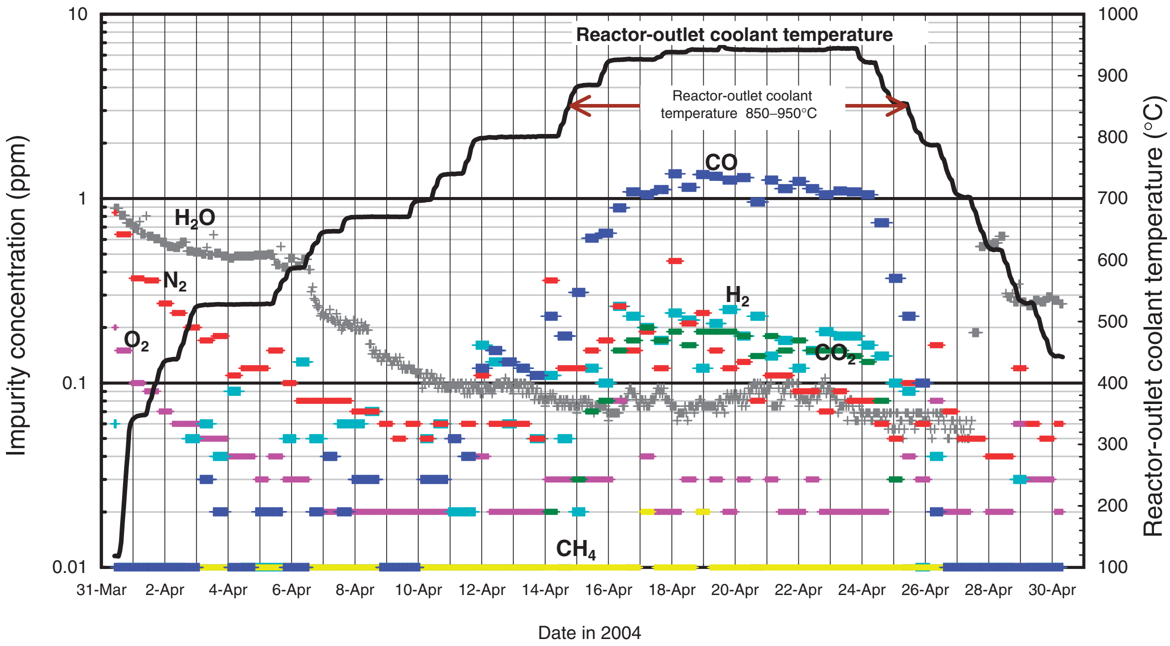

Fig. 7 Chemical impurity behaviour at the reactor-inlet

Table 4 Core internal structure temperature at the reactor power of 100% (30 MW)

Measurement point	Design value (°C)	Limited value (°C)	Measurement data (°C)
Core restraint mechanism restraint band surface	450	—	386–391
Permanent reflector block outer surface	550	—	421–471
Permanent reflector block inner surface	700	—	413–533
Bottom block upper surface	480	—	444–433
Core support plate upper surface (centre)	530	530	450
Core support plate upper surface (surrounding)	470–530	530	406–427
Seal plate upper surface	450	—	397–399
High-temperature plenum block	1050	—	742–755

Table 5 Maximum fuel temperature estimated by using the obtained data

Item	Location	Pre-evaluation[6] (°C)	Measurement data (°C)
Core-inlet coolant temperature	Centre region	399	396
	Surrounding region	405	402
Core-outlet coolant temperature	Centre region	984	991
	Surrounding region	952	954
Temperature rising at the core	Centre region	585	595
	Surrounding region	547	551
Maximum fuel temperature[a]		1,463	1,478

[a] Maximum fuel temperature is a calculated value and its normal operation limit is 1,495°C.

the rate of change in reactivity per unit thermal power. The amount of the reactivity change was calculated using the control-rod worth curve from the difference of the control-rod insertion length for each power level.

The experimental results of the power coefficient of the reactivity for the high-temperature test operation up to 30 MW are shown in **Fig. 8**. The control-rods position at the full power during this period was about 2,800 mm. The active core region is from 0 to 2,900 mm. The previous experimental results are also shown in this figure. There is no significant difference in the coefficient between the present and previous operations. Since a change of the core temperature during the high-temperature test operation with a reactor-outlet coolant temperature at 950°C was bigger than

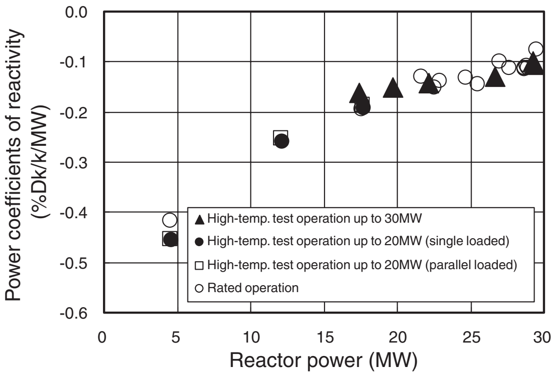

Fig. 8 Power coefficients of the reactivity

that of the rated operation mode with a reactor-outlet coolant temperature at 850°C, the power compensation reactivity of 0 to 100% of the high-temperature test operation became around 0.3% Δk larger than that of the rated operation. It can be argued that the difference of the power coefficient which is the reactivity change per power increment of 1 MW between the rated operation and the high-temperature test operation is actually negligible because the change of the core temperature was about the same in both operations.

6. Fuel and Fission Product Gases Behaviour

Fuel and fission product gases behaviour was monitored in order to evaluate the release behaviours of the fission product gases and to confirm that the levels of the released fission product gases were within their limits during the operation. The primary coolant radioactivity instrumentation of the safety protection system, the fuel failure detection (FFD) system, and the primary coolant sampling system have been installed in the primary circuit for the measurement of the primary coolant radioactivity.[7] The primary coolant radioactivity was measured continuously during this operation as shown in **Fig. 9**. Results were that, not only were all signals less than the alarm level of 10 GBq/m^3 which corresponds to 0.2% of fuel failure, but also all signals were less than the detection limit (1 GBq/m^3).

The FFD system was employed in the HTTR to prevent the additional abnormal failure of the coated fuel particles during normal operations. **Figure 10** shows the FFD signal from the No. 1 region for the reactor power during this operation. The signal varied in proportion to the reactor power, which increased linearly up to 60% of the reactor power and exponentially thereafter. Comparing with rated operation modes, FFD signals in high-temperature test operation mode took a higher value over 60% of the reactor power.

Primary coolant sampling measurement is the only way to determine the fission product gas concentrations. As results, the detected fission gas nuclides in the primary coolant were ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{133}Xe, ^{135}Xe, ^{135m}Xe, and ^{138}Xe, all of which are the same isotopes as for previous tests in rated operation mode.[7] The measured release-to-birth ratios, (R/B)s, of ^{88}Kr as a function of the reactor power are plotted in

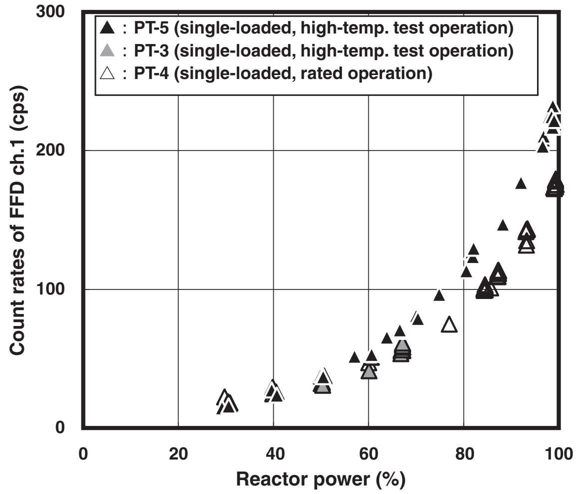

Fig. 10 Count rates of the FFD system vs. reactor power during the PT-5 operation which is the fifth phase of the rise-to-power test performed by the single-loaded and high-temperature test operation mode

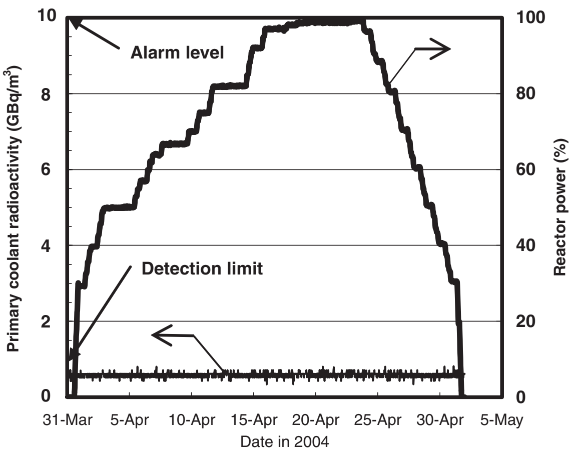

Fig. 9 Primary coolant radioactivity signals during the PT-5 operation which is the fifth phase of the rise-to-power test performed by the single-loaded and high-temperature test operation mode

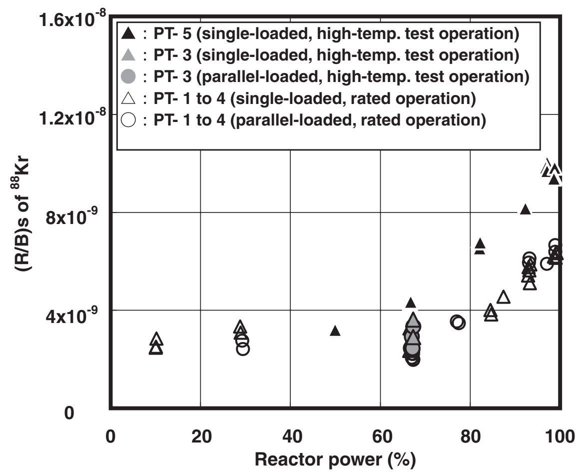

Fig. 11 Fractional release fractions of ^{88}Kr during the rise-to-power tests

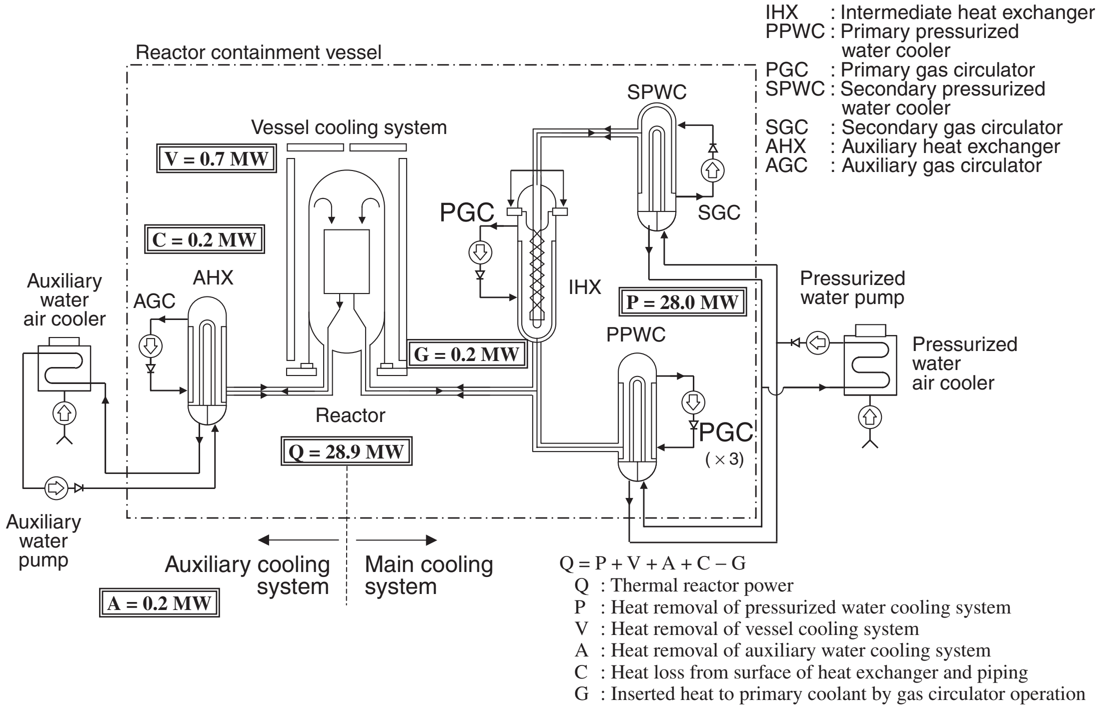

Fig. 12 Measurement result of the thermal reactor power at the calibration of neutron instrumentation

Fig. 11. In this operation, the measured fractional release at 50% of the reactor power shows the same levels as in rated operation mode, and then increase exponentially to 1.0×10^{-8} at the full power operation, which was slightly larger than in the rated operation mode, 7×10^{-9}.[7)] The measured (R/B) at the full power was three orders lower than the limitation of 5.35×10^{-4}, which corresponds to 1% fuel failure. It suggests that the measured values were within the release level by diffusion of the generated fission gas from the contaminated uranium in the fuel compact matrix, and no significant failure occurred during 950°C operation.

7. Calibration of Neutron Instrumentation System

The calibration of the neutron instrumentation system with the thermal reactor power was performed at near full power operation (the reactor power of about 97% of 30 MW). The thermal power of the HTTR is calculated with the following:

$$Q = P + V + A + C - G,$$

where Q: Thermal reactor power
P: Heat removal of the pressurized water cooling system
V: Heat removal of the vessel cooling system
A: Heat removal of the auxiliary water cooling system
C: Heat loss from the surface of the heat exchanger and piping
G: Inserted heat to the primary coolant by the gas circulator operation.

The neutron instrumentation system was calibrated to the measurement result of the thermal reactor power as shown in **Fig. 12**. The neutron instrumentation system called 'power range monitor (PRM)' is used in the range from 0.1 to 120% of 30 MW. The uncompensated ionization chamber is applied as a neutron detector for the PRM. Three detectors for the PRM are located at the symmetrical position circumferentially outside the reactor pressure vessel. The difference between the indicated value of the PRM and measured value of the thermal reactor power at the rated power operation was about 0.2% after the PRM calibration. It was confirmed that the calibration error, which was one of the important parameters in the evaluation of maximum fuel temperature, was within 2.0% taking account of the calorimetric error 1.5% of the thermal reactor power (0.2%+1.5%<2.0%).

IV. Concluding Remarks

The Japan Atomic Energy Research Institute (JAERI) achieved the rated thermal power of 30 MW and reactor-outlet coolant temperature of 950°C in the HTTR on 19 April 2004.

The reactor-outlet coolant temperature of 950°C makes it possible to extend high-temperature gas-cooled reactor use beyond the field of electric power. Also, highly effective power generation with a high-temperature gas turbine becomes possible, as does hydrogen production from water. Continuous hydrogen production by the thermo chemical IS process has already been demonstrated in bench scale tests by JAERI in August 2003. In this process, hydrogen was produced from water with no carbon dioxide emission. The achievement of 950°C will be a major contribution to the actualization of producing hydrogen from water using the high-temperature gas-cooled reactors.

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